U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

RIC 2011

NRC-Sponsored Environmentally Assisted Fatigue Research Activities

Background

General Design Criteria (GDC) 1 and 30 in Appendix A to 10 CFR Part 50 require that components that are part of the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest practical quality standards. Augmenting those design criteria, 10 CFR 50.55a, "Codes and Standards," endorses the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code for design of safety-related systems and components by requiring, in part, that components of the reactor coolant pressure boundary meet the requirements for Class 1 components in Section III of the ASME Code. Part of the design consideration is a fatigue evaluation using a fatigue cumulative usage factor (CUF) calculation in accordance with the ASME Code, Section III, fatigue design rules. To properly consider the effects of the reactor water environment on fatigue of metal components, NRC Regulatory Guide (RG) 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," and associated NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," describe the environmental fatigue multiplier (F_{en}) methodology as an acceptable approach for evaluating the effective fatigue life of components exposed to a water environments.

Over the past 10 years, evaluation for license renewal and new reactors has provided significant experience and insight on the use of the F_{en} approach and revealed the need for further refinement of this methodology, as well as solving some practical issues associated with applying the methodology in typical calculations. Hence, the NRC has initiated further research work on environmentally assisted fatigue. This poster provides an overview of these environmental fatigue efforts.

Motivation

There is a need for additional efforts by the NRC in this area to facilitate the future review of licensees' environmental fatigue evaluations associated with license renewal and new reactor applications.

There is a need for development of a transient stress evaluation software tool and a fatigue calculation software tool that satisfies ASME Code, Section III, requirements that the staff can use to perform independent assessments of environmental fatigue submittals by licensees.

There is a need for development of a revised fatigue usage factors (CUF) limit for high-energy line break (HELB) locations that includes environmental fatigue effects. The current CUF limit is 0.1 for the 40-year design life of the component without consideration of environmental effects.

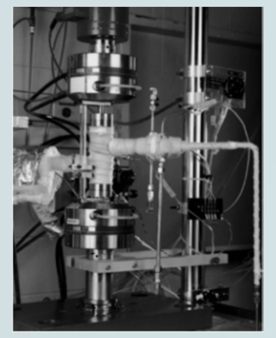
Objectives of NRC-Sponsored Research

Develop a transient stress evaluation software tool for rapidly determining thermal transient stresses in reactor components.

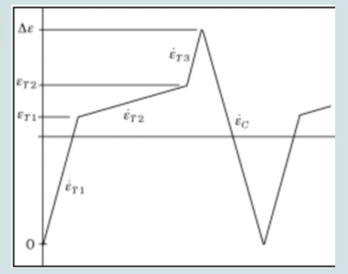
Develop an ASME Code, Section III, fatigue calculation software tool for estimating CUF in reactor components.

Develop revised CUF limit criteria for postulated HELB locations.

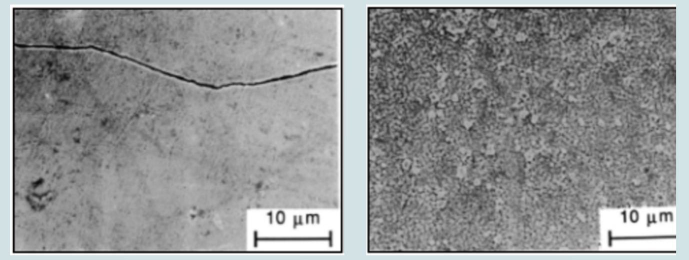
Obtain technical support from Argonne National Laboratory (ANL) to update existing environmental fatigue methodology, develop application techniques for applying the methodology, and revise RG 1.207 accordingly, if appropriate.



Autoclave system for fatigue tests in water



Loading waveform for variable strain rate tests



SEM photomicrographs of A533–Gr B steel tested in air (left) and water (right) environments at 288 °C

Current Research Efforts—ANL Investigations

ANL is currently performing additional NRC-sponsored research in the following areas:

Review the basis documents for two recently approved ASME Section III Code cases on environmental fatigue to help address unresolved issues.

Investigate strain threshold and hold-time effects on the F_{en} methodology.

Review additional available laboratory data collected over the past decade subsequent to the publication of the fatigue curves and F_{en} methodology documented in RG 1.207, investigate possible adjustments to the F_{en} methodology to clarify the approach and remove unnecessary conservatism, and revise RG 1.207 accordingly, if appropriate.

Investigate the significance of the constant in the expressions for F_{en} used in RG 1.207 (i.e., F_{en} of approximately 2 even when strain rates are high and temperature is very low) and the adequacy of the factor of 2 on strain/stress and the use of a factor of 12 on cycles.

Investigate methods for calculating strain rate, linking of transients, and cycle counting, and clarify the F_{en} methodology with respect to these items.

Collect and establish a reliable, up-to-date database of all relevant fatigue test data.

Investigation of High Energy Line Break and Cumulative Usage Factor Limit

Section 3.6.2 of the Standard Review Plan (NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"); Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements;" and Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," specify the evaluation of postulated break locations for HELB piping.

Pipe rupture must be considered to comply with the requirements of GDC 4, "Environmental and Dynamic Effects Design Bases," for the design of nuclear power plant structures, systems, and components (SSCs).

Of particular interest are the portions of these criteria that limit the CUF to a value of 0.1 for the 40-year design life of the component for the locations where breaks and cracks are not postulated.

A long-term solution for establishing HELB criteria is a rigorous probabilistic approach, such as may be provided by the Extremely Low Probability of Rupture (xLPR) Program (still under development).

In the short term (before completion of the xLPR Program), the NRC is engaged in a cooperative study with the Electric Power Research Institute (EPRI) (under an addendum to the NRC/EPRI Memorandum of Understanding) for new reactors and plants operating beyond 40 years with the intent of determining an appropriate CUF limit for up to 60 years of operation, while also including environmental fatigue effects.

The NRC's investigations and research efforts in this area are intended to align requirements for operating reactors and new reactors with consistent HELB CUF acceptance limits and to develop a technical basis that supports a revised CUF revised limit, as follows:

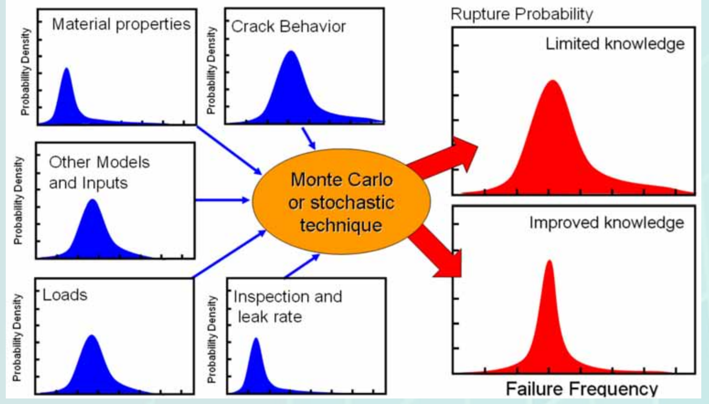
Document the available technical basis that was used to establish the 0.1 CUF limit. Survey available records for operating plants and establish a list of HELB locations.

Establish a common set of HELB locations for evaluation.

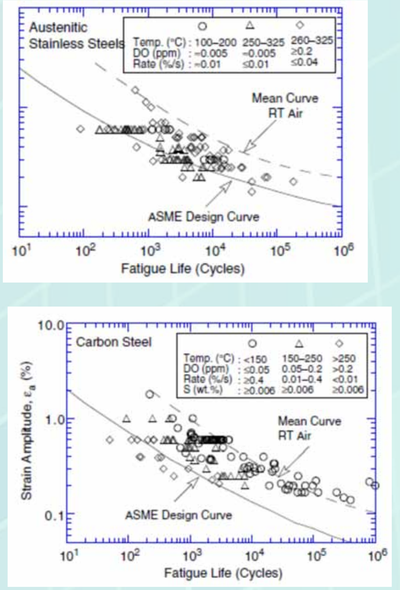
Obtain the governing CUF calculations for the common set of locations.

Perform refined calculations considering 60 years of operation, environmentally assisted fatigue (EAF), and ASME Code Subparagraph NB-3200 methodology.

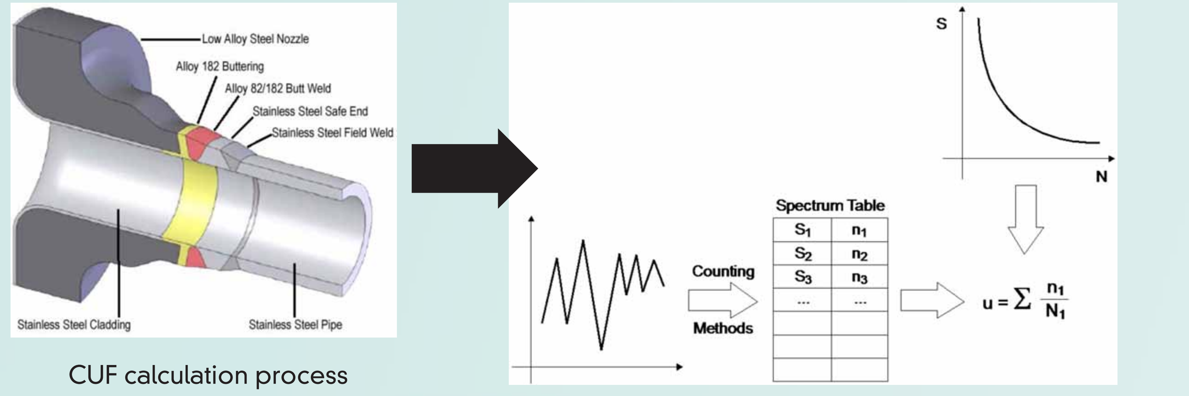
Establish the increase in CUF for all locations, and establish a revised CUF limit.



Longer term (xLPR) approach for evaluating HELB locations



Fatigue S–N data for carbon steels (upper) and stainless steels (lower) in water



CUF calculation process

ASME Code Fatigue Calculation Software Tool Development

The NRC is developing ASME Code fatigue calculation software tools for internal use, as follows:

Development of a Transient Stress Evaluation Software Tool

A "mathematical integrator" will more quickly and efficiently develop the thermal stress response to any input thermal transient, without the need to iterate a finite element model for many thermal transients.

This tool will save significant analysis time, as the finite element analysis setup for thermal transients can be significant, especially for reactor pressure vessel (RPV) components that have several dozen complicated thermal transients associated with their design basis.

Development of an ASME Code, Section III, Fatigue Calculation Software Tool

Calculating the CUF for reactor coolant pressure boundary components in accordance with ASME Code, Section III, Subparagraph NB-3200, especially for RPV components, is very laborious and requires significant experience and judgment on the part of the analyst (as highli.ht Summary 2008-30, "Fatigue Analysis of Nuclear Power Plant Components," dated December 16, 2008).

This software tool will perform CUF calculations using ASME Code Subparagraph NB-3200 methodology and will incorporate the collective expertise and judgments of NRC staff members who have expertise in this area.

Timeline

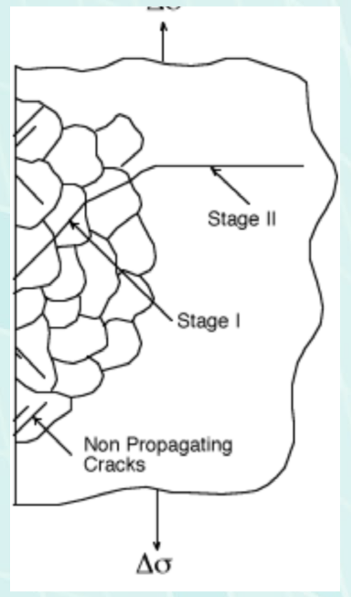
Work on further data investigations and development of fatigue software tools by the NRC and ANL—present

HELB investigation completed—September 2011

ANL data investigation completed—December 2011

Fatigue calculation tools completed—December 2011

Recommended actions, research complete—December 2011



Two stages of fatigue crack growth in smooth test specimens

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